Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/521
AUSTRALIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

A NEW TECHNIQUE FOR UP-GRADING AUSTRALIAN THORIUM RESOURCES

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Since Australia has virtually no workable deposits of thorium comparable with the oxide and silicate minerals of the Canadian uranium fields thorium produced locally must be obtained from monazite. Total reserves of this mineral are not accurately known but production has been increasing rapidly with the growth of industries processing beach sands for ilmenite, rutile and zircon. In 1962, 899 tons of refined monazite were produced, mainly from the ilmenite treatment plants of Western Australia. The remainder came from New South Wales and Queensland. Almost the whole of this production, together with accumulated reserves of concentrate, were exported to Europe and the United States as a high-grade product containing over 95 per cent of monazite.

Extraction of thorium from monazite has not been attempted in Australia on a commercial scale. However, ample facilities exist to do this economically if interest in Th-U^{233} and Th-Pu^{239} fuel cycles is maintained by the Australian Atomic Energy Commission or if other commercially attractive outlets arise. Many needlessly complicated processes have been proposed for this extraction while many "shortened" processes do not give a product of adequate purity in economic yield. A simplified "acid" process is now proposed in which a thorium concentrate is produced in a small number of stages with minimum quantities of inexpensive reagents. The yield of thorium exceeds 95 per cent. The concentrate contains an approximately equal weight of rare earths as carbonates or hydroxides. After its solution in nitric acid a suitable nitrate feed solution is provided for the tributyl phosphate solvent extraction process needed for refining thorium to nuclear grade.

Principal features of the process are (1) the quantitative co-precipitation of thorium with rare earths by means of recycled sodium sulphate solution; (2) conversion of the double sulphate precipitate to double carbonates by treatment with sodium carbonate solution; (3) selective leaching of the mixed carbonates with sulphuric or other acid to separate thorium from rare earths. The phosphate content of the residual thorium concentrate can be lowered, if required, by treating it with concentrated sodium hydroxide.

Rare earth salts are valuable by-products and it is pointed out that the choice at any location between this process and the most competitive "alkali" process may largely be determined by the comparative profitability of by-product sales and the marked price differential which favours the use of sulphuric acid in Australia.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/522
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

PRESSURE VESSEL FOR THE FIRST CZECHOSLOVAK NUCLEAR POWER STATION

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One of the basic power reactor types tested within the framework of a wide research of different prototypes in USSR and people's democratic states is the pressure vessel reactor fuelled with pin type fuel elements of natural uranium, moderated with heavy water and cooled with gas.

The basic dimensions of the reactor pressure vessel resulted from the mentioned conception and from the fixed power rating of 150 MWe. In the prescribed minimal spacing of the fuel channels large vessel dimensions and wall thickness were reached which are difficult to manufacture to weld and check. That is why the design must solve the question of optimal dimensions and designs in most troublesome parts of the vessel as the cover, the flanged joint, the conical support and above all the inlet and outlet connecting branch for the cooling gas ducting. These problems were solved theoretically and experimentally on models by means of tensometry and photoelasticimetry. Various effects on the magnitude of stress concentration in places of weakness in the cylindrical pressure vessel due to connecting branches were studied: the magnitude and shape of the hole, the number of holes on the circumference, the reinforcement of the hole with connecting branch and the local transition modification of the branch into the pressure vessel wall.

In welded pressure vessels there is the danger of sudden brittle fractures.

For this reason the informative bending tests of large bodies with notches were carried out. The result shows the dependence of strength upon magnitude and notch effect. The analysis of deformation characteristics and specification of fast plastic and fast brittle fracture ideas are mentioned.

Furthermore a project was worked out for systematic testing of large bodies and specimens as thick as the real pressure vessel wall. Tests of these specimens are carried out on large load apparatus of 6000 t force. Results of some tests of large bodies completed to this time were studied as well as their value with respect to the character of boundary condition of the reactor vessel.

Critical analysis of the importance of tests on small specimens and results of tests according to transition temperature and crack arrest temperature was carried out.

The pressure vessel will be manufactured from low carbon, fine grain, manganese structural steel produced in non-aging quality in open-hearth furnaces and cast at atmospheric pressure into ingots weighing 43 t.

Both shaped forgings of semi-products with branches, and slabs from which plate semi-products are pressed are made from ingots by free forging under a 12 000 t press. From the semi-products segments are die pressed under a 12 000 t press, from which parts of the pressure vessel are welded by electroslag welding. The complete units from the finished parts, i.e. the cover and bulk of the pressure vessel, are made by circumferential electroslag welding, semi-automatic welding under protective atmosphere of CO_2 and manual welding. The pressure vessel is stress relieved in a single-purpose electrical resistance furnace.

Before welding it was necessary to carry out extensive experimental work to attest the weldability of steel, to develop suitable additional materials for all welding methods to master the technology of welding and preheating, to verify the magnitude of stress and strain, to find suitable post-welding heat treatment etc. These tests were carried out not on plates only, but on full scale parts as well.

The whole technological cycle of the pressure vessel manufacture from the production of steel till the testing of the welded and heat treated pressure vessel, includes a sequence of operations affecting the quality of the base material and of the welded joints. For this reason inter-operational take-over tests are inserted into various phases of manufacture which have the purpose to identify the defective piece in time.

To test the development manufacturing technology and to draw conclusions as to the safety of the pressure vessel, the manufacture of the shortened model of this vessel in scale 1:1 is now nearing the end. The model is without the plain cylindrical portion and the ring with branches for the cooling gas inlet. The model will be put to strength and functional tests.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/523
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

PROBLEMS OF DEVELOPMENT AND CONSTRUCTION OF THE FIRST CZECHOSLOVAK NUCLEAR POWER STATION

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A nuclear power station with 150 MWe gross power is being built in Czechoslovakia, furnished with gas cooled heavy water moderated reactor fuelled with natural uranium metal. Description of this power station was given at the second Geneva conference (A/Conf. 15/P/2092). The station is the joint achievement of the Czechoslovak-Soviet cooperation in the field of the peaceful uses of nuclear energy. The adopted conception of construction of the full scale power station without relevant experimental reactor requires verification of all important parts of the station by scientific-technical works in both nuclear and classical parts. The centre of these works is in the development, manufacture and functional verification of the station main part models in the scale 1:1. Financial and especially time pressure in respect of these works of course lead to some delay in the construction of the power station. The presented paper deals with works carried out in Czechoslovakia.

The main attention was devoted to the reactor pressure vessel development. The material was carefully chosen and the technology, especially welding, was tested. Problems of experimental strength were solved on many model installations from which the shortened model 1:1 is nearing the completion.

With respect to the high power rating and short time of the fuel exposition incidental to it the on-load fuel exchange has become a necessity. Great attention was therefore devoted to the development and design of the charge-discharge machine. Both separate parts and the whole prototype of this machine are functionally tested.

Mechanisms of all control and scram rods were subjected to versatile tests on a special stand that permitted experiments with model mechanism in the scale 1:1 at the working temperature and pressure of the cooling CO_2. Moreover the reliability and quality of the control system was tested on a prototype control loop with electronic model of the reactor kinetics.

Technology and operational ability of the steam generators were tested on fuel-scale model of one section, great attention being devoted to the tightness. The effects of temperature shocks in piping due to little thermal capacity of the reactor at scramming were investigated.

The applicability of unified parts of a classical power station turbine for working conditions in a nuclear power station were discussed especially with respect to greater moisture content of the steam.

Information and documentary shots of the site show the advanced state of the power station construction.

Development possibilities of this reactor type are under discussion.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/524
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Submitted in:
ENGLISH and RUSSIAN

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Abstract

THE INFLUENCE OF d - c ELECTRIC FIELDS ON THE HEAT TRANSFER TO COOLING CO_2 AT HIGHER PRESSURES IN A NUCLEAR REACTOR

F. Berger and L. Derian

The cooling gas which flows through the core of a nuclear reactor is partially ionized by radiation. A flow of ions perpendicular to the gas flow may be caused by the application of an electric field. As the concentration of ions in the gas volume is relatively high, it is possible to affect in this way the whole gas-stream and especially the heat-transfer from fuel elements to the gas. At the Second Geneva Conference we reported about experiments carried out with an electrically heated fuel element model cooled by air of atmospheric pressure. The possibility of heat transfer intensification by the application of an electric field in nuclear reactors was demonstrated by these experiments.

A series of experiments has since been performed in the gas-cooled loop of the Czechoslovak reactor VVRS, first with an electrically heated model and later with a rodtype uranium fuel element. The paper gives mainly the results of these last experiments, in which the pressure of the cooling carbon dioxide was changed from 5 to 30 kp/cm^2, and the voltage between the electrodes (one of which being formed by the uranium fuel element and the other by the channel wall) was changed from 0-30 kV. In addition to the influence of pressure and electric field strength the dependency of the effect on gas velocity, density of gas ionization and the polarity of the heat-exchanging surface was studied. The experiments resulted in the recognition of some new facts as well as in the broadening of previous knowledge.

It was established that by the action of a d - c electric field both an increase ("positive effect") and a decrease ("negative effect") of heat transfer to the gas may be obtained.

The influence of the electric field on heat transfer from the fuel element to the partially ionized gas differs principally whether the fuel element works as the cathode or as the anode.

With negative polarity of the fuel element the relative increase of heat transfer is highest at low pressures. The value of the relative heat transfer increase grows with increasing field strength and increasing density of gas-ionization. It decreases with increasing gas velocity and pressure. At higher pressures the positive effect changes to the negative one; the pressure at which this change occurs depends on the other determinant parameters.

With positive polarity of the fuel element no influence of the electric field on heat transfer was observed at atmospheric pressure. At lower gas pressures (5 kp/cm^2) a negative effect occurs. By raising the pressure further the effect may be transformed into the positive one. At higher pressures (15 kp/cm^2 or more) a considerable heat transfer increase may be obtained. The value of relative heat transfer increase grows again with increasing field strength and ionization density. It decreases with increasing gas-velocity, but contrary to the previous case it increases (in the studied range of pressures) with increasing pressure.

The maximum relative increase of heat transfer reached during these experiments amounted to about 50%. The maximum heat transfer decrease (negative effect) was about 15%. A more detailed description of the experiments, the magnitude of the effect as a function of the main parameters and an interpretation of the effect will be given in the paper.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/525
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

ABOUT SOME DYNAMIC PROPERTIES OF A NUCLEAR POWER STATION WITH A HEAVY-WATER MODERATED, GAS-COOLED REACTOR

P. Štirský and J. Hulovec

This paper deals with the first Czechoslovak nuclear power station. Dynamic properties of the main technological equipment are briefly analyzed and the control system ensuring operation at the given power level of the turbo-generator is described in the introduction.

In the further part of this paper the authors analyze some transients of the atomic power station, provoked by the failing of the technological equipment, and of the regulating system.

In case of accidental shut-down of the reactor by the profecting system, there appear considerable changes in temperatures and pressures in the primary circuit. This provokes an unfavourable stress of structural parts of the reactor. There were analyzed different methods how to decrease and restrict the temperature shocks, either by the arrangement of design or by the quick restriction lowering the coolant flow.

The failing of one or two main circulation blowers is from the operation standpoint a probable failure, resulting in a considerable loss of the coolant flow in the reactor. The reactor regulating system specified for a normal operation is not able to settle these failures with allowable exceeding of temperatures of the fuel canning. Some results of measures for limiting this type of failure are introduced here.

An accidental cutting off of the turbine does not signify for the reactor any larger disturbance due to steam by-pass, designed for the full power. Some results for the regulation of the by-pass in the case of such a failure are briefly mentioned.

Even the influence of a failure of one of the twelve tubing loops upon the performance of the regulating system was taken into consideration.

A failure of the reactor outlet gas temperature regulator, resulting in the loss of the regulator output signal is the main failure of the regulating system. This failure leads to an instability of the whole regulated object. Measures for suppression of such a failure are analyzed.

The dynamic properties of the nuclear power station during non-nominal reactor power levels are shortly explained. In the conclusion there are appreciated regulating properties of this type of a nuclear power station in the electric power system.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/526
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

DIFFUSION BONDING OF URANIUM WITH MAGNESIUM BY MEANS OF INTERLAYERS

V. Kraus

Due to mutual solid state immiscibility of uranium and magnesium the bonding of these metals without a high joint deformation is impossible.

The present report describes the results of experimental diffusion welding of uranium with magnesium by means of thin intermediate layers. For these interlayers were chosen metals with low thermal neutron absorption cross-sections: zirconium, aluminium, nickel and copper; their metallurgical properties permit to assume a diffusion interaction with uranium as well as magnesium. Zirconium and aluminium were used in the form of foils, nickel and copper and were deposited electrolytically on uranium. The interlayer thickness was usually 15μ. The samples (pellets) of uranium - interlayer - magnesium were welded by diffusion at a constant initial compression and at various temperatures and times of annealing. The weld quality was investigated by measuring the sliding weld strength (at ambient and elevated temperatures) and by the methods of optical metallography.

During the interdiffusion of zirconium with uranium and magnesium in the range from 500°C to 590°C limited solid solutions of uranium and magnesium in zirconium were predominantly formed. The sliding weld strength after diffusion annealing attained values up to 300 kg/cm^2 or even more (at ambient temperature). This weld strength was conserved even after a long-time heating (810 hours) at 500°C. The strength of the U-Zr-Mg weld at a temperature of 500°C attained approximately 45 kg/cm^2. This strength practically also did not change with the long-time annealing mentioned above.

Aluminium interlayers yielded different results according to the thickness of the foil used. During a diffusion treatment in the range of 410-500°C the growth of intermetallic compounds of uranium with aluminium is

very intensive. An aluminium interlayer of a thickness of 8μ reacts throughout with uranium sooner than any considerable interaction with magnesium takes place. The welds produced in this way showed a sliding strength of nearly 500 kg/cm^2, which did not change with a long-time heating. The strength at 500°C attained in this case values up to 70 kg/cm^2. Thicker aluminium interlayers (40 to 80μ) from a number of solid solution and intermetallic phases layers at the interface with magnesium; at the welding temperature over 437°C an eutectic alloy layer is formed. The weld strengths are determined here predominantly by the properties of these phases. The undereutectic diffusion layers showed at room temperature a sliding strength of approximately 130 kg/cm^2, the eutectic alloy over 500 kg/cm^2. At the temperature of 500°C, however, the weld strength is minimal due to the presence of a liquid phase. After a long time heating at 500°C considerable part of aluminium which is not bound on uranium enters the solid solution with magnesium without affecting essentially the weld strength.

The characteristics of uranium welds with magnesium using nickel and copper intermediate layers are very similar. At a short-time diffusion heating below the eutectic temperature (507°C for the Ni-Mg system and 485 for that of Cu-Mg) diffusion layers were obtained with a sliding strength of approximately 200 kg/cm^2. At a supereutectic temperature annealing predominated at the interface an eutectic alloy, which showed the strength up to 400 kg/cm^2 at room temperature. During a long-time heating at about 500°C this strength still somewhat improves, but the eutectic alloy quickly penetrates along the magnesium grain boundaries and thus seriously deteriorates the properties of the canning material.

These results show that sufficiently strong and long-time stabile magnesium to uranium diffusion bonds may be attained by means of thin aluminium (approximately 8μ) and zirconium intermediate layers.

The experiments carried out have proved the dependence of the bond strength on temperature and time of diffusion treatment. Assuming the bond strength dependence on the diffusion transfer of a certain amount of mass necessary for forming a cohesive intermediate region between the joined crystal lattices a relation has been derived which describes the time-temperature dependence of the bond strength. The validity of this relation was checked experimentally for the sliding strength of the U-Zr-Mg bonding, which could be investigated in broader ranges of diffusion treatment temperatures and times.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/527
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

SOME HEAT CONDUCTION THEORETICAL PROBLEMS IN REACTOR FUEL ELEMENTS

J. Schmid

A heat conduction boundary problem of nuclear reactor fuel elements with temperature dependent thermal conductivity and with nonlinear boundary conditions of the third and fourth type, i.e. with temperature dependent heat-transfer coefficient or with temperature dependent contact resistance on the interface of two media, is solved.

The heat transfer equation in cylindrical geometry

(1) $$\left[\frac{1}{r}\frac{\delta}{\delta r}\left(\lambda r\frac{\delta t}{\delta r}\right)+\frac{1}{r^2}\frac{\delta}{\delta\varphi}\left(\lambda\frac{\delta t}{\delta\varphi}\right)\right]=g(r,\varphi)$$

was transformed using the expression

(2) $$\lambda(t)=\lambda_o+\lambda_1(t)$$

into the form

(3) $$-\lambda_o\left[\frac{\delta^2 t}{\delta r^2}-\frac{1}{r}\frac{\delta t}{\delta r}+\frac{1}{r^2}\frac{\delta^2 t}{\delta\varphi^2}\right]=g(r,\varphi)+\lambda_1(t)\left[\frac{\delta^2 t}{\delta r^2}+\frac{1}{r}\frac{\delta t}{\delta r}+\frac{1}{r^2}\frac{\delta^2 t}{\delta\varphi^2}\right]+\frac{d\lambda_1}{dt}\left[\left(\frac{\delta t}{\delta r}\right)^2+\frac{1}{r^2}\left(\frac{\delta t}{\delta\varphi}\right)^2\right].$$

Supposing the contribution of the nonlinearities (i.e. the second and third terms on the right side of equation (3)) is small, we can find the solution using an iterative process. After rearrangement of the equation (3) we obtain

(4) $$\frac{\delta^2 t^{(m)}}{\delta r^2}+\frac{1}{r}\frac{\delta t^{(m)}}{\delta r}+\frac{1}{r^2}\frac{\delta^2 t^{(m)}}{\delta\varphi^2}=-g^{(m)}$$

where

$$g^{(o)}=\frac{1}{\lambda_o}\,g(r,\varphi) \qquad (m=o)$$

$$g^{(m)}=\frac{1}{\lambda_o}\left\{g(r,\varphi)-\lambda_1(t^{(m-1)})g^{(m-1)}+\frac{d\lambda_1}{dt}\Big|_{t=t^{(m-1)}}\left[\left(\frac{\delta t^{(m-1)}}{\delta r}\right)^2+\frac{1}{r^2}\left(\frac{\delta t^{(m-1)}}{\delta\varphi}\right)^2\right]\right\}$$

$$(m=1,2,3,\ldots).$$

The solution of the equation has the form

$$(5)\quad t^{(m)} = A_o^{(m)} + B_o^{(m)} \ell_{gr} + {}^{c}Z_o^{(m)} + \sum_{n=1}^{\infty} [(A_n^{(m)} r^n + B_n^{(m)} r^{-n} + {}^{c}Z_n^{(m)}) \cos n\varphi +$$

$$(C_n^{(m)} r^n + D_n^{(m)} r^{-n} + {}^{s}Z_n^{(m)}) \sin n\varphi]$$

where

$$\ell_{Z_n^{(m)}} = -r^n \int_{r_o}^{r} r'^{-(2n+1)} \int_{r_o'}^{r'} (r'')^{n+1} \ell_{\overline{g}_n^{(m)}}(r'')\, dr''\, dr' \qquad \ell = \begin{cases} c & (n=0,1,2,\ldots) \\ s & (n=1,2,3,\ldots) \end{cases}$$

and $\ell_{\overline{g}_m^{(m)}}(r)$ are the Fourier coefficients of the function $g^{(m)}$.

The boundary conditions (with small nonlinearities) can be expressed in a similar way; for example the boundary condition of the third type, which has the form

$$(6)\qquad -\lambda(t) \left.\frac{\delta t}{\delta r}\right|_{r=R} = \alpha(\varphi, t, t_R)\, [t(R,\varphi) - t_R(\varphi)]$$

can be written using relation (2) and the expression

$$\alpha(\varphi, t, t_R) = \phi(\varphi)[1 + F(t, t_R)]$$

as

$$(7)\quad \lambda_o \left.\frac{\delta t^{(m)}}{\delta r}\right|_{r=R} + \phi(\varphi)[t^{(m)}(R,\varphi) - t_R(\varphi)] =$$

$$-\phi(\varphi) F(t^{(m-n)}, t_R)\, [t^{(m-1)}(R,\varphi) - t_R(\varphi)] - \lambda_1(t^{(m-1)}) \left.\frac{\delta t}{\delta r}^{(m-1)}\right|_{r=R}$$

The solution of equation (7) can be found by a similar iterative process.

The boundary conditions varying along the perimeter were expanded in Fourier series and the boundary problems were solved using matrix calculation. External boundary conditions (the first, second, third ones) may be written in the form

$$_k\underline{M}\ \underline{C}^{(m)} = {}_k\underline{F}^{(m)} \qquad (k = 1,2,3)$$

and the conditions on the interface in the form

$$(8)\qquad \underline{C}_{\ell}^{(m)} = {}_4\underline{M}_{\ell-1,\ell}\ \underline{C}_{\ell-1}^{(m)} + {}_4\underline{F}_{\ell-1,\ell}^{(m)}$$

where $\underline{M}$ are matrixes, $\underline{F}$ vectors and $\underline{C}$ vectors whose components are integration constants A_o, B_o, A_1, B_1, C_1, D_1, ... The problem was solved also for axially

non-symetrical case (for a small eccentricity). The transformation of the co-ordinate system is of a similar form as the equation (8).

Solving the boundary problem for the annular region consisting of L annuli, it is necessary first to eliminate the integration constants of the intermediate annuli using the relation

$$\underline{C}_{\ell}^{(m)} = {}_4\underline{M}_{1,\ell}\,\underline{C}_1^{(m)} + {}_4\underline{F}_{1,\ell}^{(m)} \tag{9}$$

where

$${}_4\underline{M}_{1,\ell} = {}_4\underline{M}_{\ell-1,\ell} \cdot {}_4\underline{M}_{\ell-2,\ell-1} \cdots {}_4\underline{M}_{2,3} \cdot {}_4\underline{M}_{1,2}$$

$${}_4\underline{F}_{1,\ell}^{(m)} = {}_4\underline{M}_{\ell-1,\ell}\left\{{}_4\underline{M}_{\ell-2,\ell-1}\left[{}_4\underline{M}_{\ell-3,\ell-2}(\ldots) + {}_4\underline{F}_{\ell-3,\ell-2}^{(m)}\right] + {}_4\underline{F}_{\ell-2,\ell-1}^{(m)}\right\} + {}_4\underline{F}_{\ell-1,\ell}^{(m)}$$

and then to solve the equation system

$$\begin{cases} k_1\underline{M}_1\,\underline{C}_1^{(m)} = k_1\underline{F}_1^{(m)} \\ k_L\underline{M}_L\,\underline{C}_L^{(m)} = k_L\underline{F}_L^{(m)} \\ \underline{C}_L^{(m)} = {}_4\underline{M}_{1,L}\,\underline{C}_1^{(m)} + {}_4\underline{F}_{1,L}^{(m)} \end{cases} \tag{10}$$

After eliminating $\underline{C}_L^{(m)}$ we solve $\underline{C}_1^{(m)}$ and then $\underline{C}_{\ell}^{(m)}$ using the expression (9).

It may be expected that the iterative process will converge, if non-linearities caused by respecting the temperature dependence of the thermal conductivity, the heat-transfer coefficient and the contact resistance are small.

If the thermal conductivity, the heat-transfer coefficient and the contact resistance are temperature independent, it is sufficient to find the solution of the equation (4) for the given boundary conditions as the first step of the iterative process.

The method was applied to some problems in calculation of hot channel factors.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/528
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

EXPERIMENTAL LOOP OF THE REACTOR KS-150

F. Klik, Zběhlík, Sklenička and Velechovský

The reactor design and the project of the first Czechoslovak nuclear power plant are being solved in such a way as to permit building in channels, cooling, circulating and auxiliary facilities of the experimental loop for the examination of new types of fuel elements in an individual circulating circuit.

The loop channel in the form of a Field tube will be built into the reactor periphery technology channel to replace a normal fuel element. The building in or loop channel removing assumes reactor stopping and gas coolant outletting but it does not require removing the cover of the pressure vessel. The reactor and power plant will be put into operation without the experimental loop. The building in the channel, including the loop equipment, is planned to be carried out after the initial operation will be started. Having ended the examination programme the loop channel may be removed from the reactor again and the normal technological channel with the fuel element inserted to its place.

The experimental loop has its individual circuit which is completely separated from that of the power plant. Only if carbon dioxide is used the gas system of the power plant will be used for filling and emptying the loop circuit. The basic equipment of the main circuit (blowers and gas-water heat exchangers) is doubled and sized for the full power, normally both units being operated at the same time so that loop cooling is not interrupted if one unit fails. In addition, a regeneration gas-gas heat exchanger and a filter for coolant purifying are built in the main circuit.

Generally transport technological facilities of the power plant will be used for handling both fresh and irradiated fuel elements examined. This

facility permits to remove the element or put it into the loop channel at full power, i.e. without shutting down the reactor and without interrupting the circulation in the main loop circuit.

After ending the irradiation the fuel element examined will be cooled in the reactor cooling zone and later in a special gas vessel. Its cooling will be assured by an auxiliary circuit equipped with particular blowers and heat exchangers.

The loop failure conditions were analyzed and such modifications have been performed in the project that it could not practically affect the reactor operation and would maximally limit the necessity of reactor shut down due to loop operation.

The loop will permit examining large fuel elements (diameter up to 65 cm, length up to 3.7 m) in a high neutron flux and under high thermal loads. The thermal loop output will attain up to 1.75 MW, coolant consumption up to 23 t/h and pressure gradient in the main circuit up to 20 kg/cm^2.

The channel design and loop project, its essential parameters and experimental capacity will be described in the report and the effect of this design on reactor and power plant operation are analyzed.